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PEACH BOTTOM 2 TURBINE TRIP 2 SIMULATION

by TRACE/S3K COUPLED CODE

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ABSTRACT

A coupling between the TRACE system thermal-hydraulics code and the SIMULATE-3K (S3K) three-dimensional reactor kinetics code has recently been developed in a collaboration between the Paul Scherrer Institut (PSI) and Studsvik. In order to verify the coupling scheme and the coupled code capabilities the NEA/OECD Turbine Trip benchmark was simulated. The core/plant system data were taken from the benchmark specifications while the nuclear data were generated with Studsvik's lattice code CASMO-4 and core analysis code SIMULATE-3. The comparison with the experimental data shows that the TRACE/S3K code reproduces well the main transient parameters, namely, the pressure wave propagation, void collapsing and core power response.

Key Words: Turbine Test, Coupled Code, Pressure Wave, Void Collapse

1. INTRODUCTION

The TRACE/S3K thermal hydraulics and three dimensional reactor kinetics coupled code was developed to consolidate the performance of two principal reactor analysis transient codes used at PSI by the STARS (Safety research related to Transient Analysis of Reactors in Switzerland) [1] project for supporting the existing Swiss Nuclear Power Plants (NPPs) . The first code is the two group nodal kinetic code SIMULATE3K (S3K) [2] and the second one is the best-estimate thermal-hydraulic system code TRACE [3]. The motivation at the basis of the development of the TRACE/S3K coupled code is the continuing development of the capability to perform best-estimate simulation of Light Water Reactors (LWRs) transients where strong coupling between the core neutronics and the plant thermal-hydraulics and/or asymmetrical power generation take place, e.g. during ATWS transients.

The S3K and TRACE codes were intensively verified and validated by the developers and the user community. However, additional validation and verification is needed for the coupled TRACE/S3K code. An independent verification matrix was proposed to confirm the coupled code capability and accuracy. It includes PWR and BWR benchmarks as well as comparison with the available Swiss NPPs data. The NEA/OECD Peach Bottom 2 Turbine Trip Test 2 (PB2 TT2) [4], which was previously simulated by means of the S3K code standalone [5] and by the coupled TRAC-M/PARCS code [6] (TRAC-M is a predecessor of TRACE), was selected as a BWR benchmark for testing the coupled TRACE/S3K code because it is a dynamically complex event for which neutron kinetics in the core was strongly coupled with thermal-hydraulics in the reactor primary system. At PSI, the PB2 TT2 benchmark was previously analyzed with the CORETRAN and RETRAN-3D codes [7] and since these codes are now being replaced by S3K and TRACE, it is considered appropriate to use the PB2 analysis as the first validation case of the coupled TRACE/S3K system for BWR applications.

The PB2 TT2 was performed at the Peach Bottom-2 BWR/4 Nuclear Power Plant (NPP) prior to shutdown for refueling at the end of Cycle 2 in April 1977. The test was selected for the benchmark problem to investigate the effect of the pressurization transient, (following the sudden closure of the turbine stop valve) on the neutron flux in the reactor core. In a best-estimate manner the test conditions approached the design basis conditions as closely as possible. The actual data were collected, including a compilation of reactor design and operating data for Cycles 1 and 2 of PB2 and the plant transient experimental data.

A brief description of TRACE/S3K coupled code is given in section 2. The test description and corresponding model nodalization are discussed in section 3, while section 4 provides the best-estimate TRACE/S3K results for the base case and comparison with the measured data. In addition, the results of a sensitivity study on the coupling technique and the number of channels used are presented in section 5.

2. DESCRIPTION OF TRACE/S3K COUPLED CODE

The S3K code is currently being established as principal 3-D kinetic solver within the STARS project at PSI and in that context, an assessment of the code for a wide-range of core transients has been launched (e.g. [9]). In parallel, the NRC-sponsored best-estimate thermal-hydraulics code TRACE was adopted to replace in a consolidated manner, the formerly used tools (e.g. TRAC-BF1, RELAP and RETRAN) for system transient analyses of the Swiss reactors. To support this migration, substantial assessment/validation efforts have been undertaken and continue to be carried out (e.g. [10], [11]). On the basis of the maturity achieved through this assessment, a coupling between the two codes was considered as an evident next step. In that context, although the TRACE code integrates the advanced neutronic PARCS solver, the main reason to couple it with S3K was to allow for fully consistent core models between the transient and static tools noting that at PSI, a CASMO-4/SIMULATE-3 based methodology is employed for the steady-state analyses of the Swiss reactors [12].

2.1. Neutron Kinetic Code S3K

The S3K code is a best-estimate nodal reactor analysis tool that employs advanced core neutronics coupled with detailed thermal-hydraulic channel models. Faithful modeling of assembly-by-assembly neutronic effects, including assembly pin power reconstruction, permits application of S3K to a wide class of LWR core transients.

Reference [5] describes in details the S3K neutronics. A brief description is provided in what follows for completeness. The neutronic model used in S3K solves the transient three-dimensional, two-group neutron diffusion equations, including a six group model for delayed neutron precursors. S3K tracks dynamically nodal concentrations of fission products and accounts for the neutron sources due to spontaneous fissions, alpha-n interactions from actinide decay, and gamma-n interactions from long-term fission product decay.

The basic spatial integration model of S3K is formed via transverse integration of the 3-D equations separately over each spatial direction. This procedure creates an equivalent set of three one-dimensional equations coupled via a transverse leakage term. The flux distribution is expanded in terms of fourth-order polynomials (or analytical functions) in each direction and thus the spatial gradient of the flux can be analytically represented by a third-order function. This procedure yields the spatial difference equations for the two-group neutron flux. These difference equations also include assembly discontinuity factors (ADFs), which take into account the fact that adjacent assemblies may contain significant material heterogeneities. ADFs are generated from first principles by the lattice physics code CASMO as part of the core design process and are stored in the basic two-group data library. The frequency transform method is used for the time integration of the transient diffusion equations. This method separates the flux into two components, one with a pure exponential time dependence, and the other with primarily spatial (and weak temporal) dependence. The time integration of the diffusion equations is performed using backward differences

The knowledge of the intra-nodal flux and power distributions within each node can be used to compute the pin-by-pin power for every axial level of every fuel pin in the core.

2.2. System Code TRACE

TRACEv5.0 [3] is the latest in a series of advanced, best-estimate reactor system codes developed by the U.S. Nuclear Regulatory Commission (with the involvement of Los Alamos National Laboratory, Integrated Systems Laboratory (ISL), The Pennsylvania State University (PSU) and Purdue University) for analyzing transient and stationary neutronic/thermal-hydraulic behaviour of Light Water Reactors (LWRs). The code is a result of a consolidation of the capabilities of previous USNRC supported codes, such as TRAC-PF1, TRAC-BF1, RELAP-5 and RAMONA. The most important models of TRACE include multidimensional two-phase flow, non-equilibrium thermodynamics, generalized heat transfer, reflood, level tracking and reactor kinetics.

The set of coupled partial differential equations, together with the necessary closure relationships, are solved in a staggered (momentum solved at cell edges) finite difference mesh. Heat transfer is treated semi-implicitly, while the hydrodynamic equations (1, 2 and 3 Dimensional) make use of a multi-step time differencing scheme (SETS) that allows the material Courant limit to be violated, thus resulting in large time step sizes for slow transients, and fast running capabilities. The system of coupled non-linear PDEs is solved by means of a Newton-Raphson iterative method, which results in a set of linearized algebraic equations in pressure, whose results is obtained by direct matrix inversion. A full two-fluid (6-equations) model is used to evaluate the gas-liquid flow, with an additional mass balance equation to describe a non-condensable gas field, and an additional transport equation to track dissolved solute in the liquid field.

2.3. TRACE/S3K Coupled Code

Typically, the TRACE core thermal-hydraulics (TH) nodalization does not include a flow channel for each fuel assembly, but only 6 to 50 effective flow channels. These flow channels comprise from a few up to 100 fuel assemblies each. Therefore, the coupling of S3K and TRACE must provide fuel assembly based TH parameters to S3K for thermal feedback effects during a transient. The only firm requirement for the nodalization in the TRACE core region is that it must contain the same number of axial nodes in the active fuel region as the S3K model. A brief description of the linkage between TRACE and S3K follows.

The linkage is a direct, explicit coupling of the two codes on a synchronous time-step basis. The coupling provides a method of executing the S3K three-dimensional neutronics using the Nuclear Steam Supply System (NSSS) boundary conditions calculated by the TRACE thermal hydraulics code. It allows the S3K calculated total core power and core power distributions to “drive” the TRACE system model core.
Detailed calculations from the component codes result in a methodology for analyzing limiting transients such as steam line breaks, rod drops/ejections, and ATWS scenarios. These transient events require detailed three dimensional core data and information about the behavior of NSSS components, such as the separators, pressurizer, steam generators, and steam lines. A coupled analysis of these transients is important because the core behavior is closely tied to the NSSS system.

The thermal hydraulic conditions in the core and plenum regions are passed to the S3K model that performs a calculation of detailed core power, which is then passed back to the TRACE model to use for the next time step. There are three different coupling options that are available for the linkage between TRACE and S3K, “plenum”, “flat“, and “nodal“ respectively. Each of these options is described in the following paragraphs.

The “plenum” coupling option utilizes the S3K thermal-hydraulics calculation module for the core section. The inlet flow and enthalpy to the core and the exit pressure in the upper plenum is provided by the TRACE model for each core channel. S3K will use this data to perform its own thermal-hydraulic calculations in the core region. Each fuel assembly in the core is modeled separately and a common bypass channel is used for all intra- and inter bypass flows. More details concerning the S3K thermal-hydraulics model are given in Ref. [2]. These thermal-hydraulic results are only used to provide feedback values on a nodal basis for the cross section evaluation. The resultant power distribution is then collapsed back to the coarse core nodalization used by the TRACE model. This option is under implementation at the moment.

The “flat” coupling option does not utilize the S3K thermal-hydraulics calculation module. Each fuel assembly in a TRACE channel receives the same fuel temperature, coolant density, and boron concentration at a given axial plane from the TRACE calculation. This option is very robust, but it will approximate the accurate radial power distribution (especially for the hot assemblies or controlled assemblies) unless a large number (>100) of TRACE channels are modeled.

The “nodal” coupling option is a variation of the “flat” option. Once again, the S3K thermal-hydraulics calculation module is bypassed. However, an estimate of the true three-dimensional density and fuel temperature distributions is made utilizing the current nodal powers. The fuel temperature is estimated from the coarse value calculated by TRACE using a weight factor, evaluated as the ratio of the nodal power to the average power in the channel in that plane. The density for a given fuel assembly is calculated using a simple enthalpy rise calculation and the same weight factor described for the fuel temperature calculation. The density calculation also includes a normalization step that preserves the mass of liquid for each TRACE channel.

3. PB2 TT2 DESCRIPTION AND MODELS INVOLVED

The PB2 TT2 starts with a sudden closure of the turbine stop valve (TSV) followed by the opening of the turbine bypass valve. From a fluid-flow phenomena point of view, pressure and flow waves play an important role during the early phase of the transient. This is because rapid valve actions cause sonic waves, as well as secondary waves, which are generated in the pressure vessel. The pressure oscillation generated in the main steam piping propagates with relatively little attenuation into the reactor core. The induced core pressure oscillation results in changes to the core void distribution and fluid flow. The magnitude of the neutron flux transient in the BWR core is affected by the initial rate of pressure rise (caused by the pressure oscillation) and has spatial variation. The detailed description of the test is available in Ref. [4].

The simulation of the power response to the pressure pulse and subsequent void collapse requires a 3-D core modeling supplemented by a 1-D simulation of the remaining reactor coolant system. A TT transient in a BWR-type reactor is considered one of the most complex events to be analyzed because it involves the reactor core, the high pressure coolant boundary, associated valves and piping in highly complex interactions with rapidly changing variables. As mentioned earlier, the transient begins with a sudden TSV closure that initiates a pressure wave in the main steam system, which is quickly transmitted to the reactor pressure vessel. While the TSVs are closing, the bypass system valves are designed to open, which allows for steam release and, thus, pressure relief. Safety relief valves (SRVs) eventually begin to open at pre-established set points, providing additional pressure relief. The pressure wave requires a detailed nodalization modeling of the steam system and its associated valves to correctly capture the timing effects and pressure wave magnitude. This assures the availability of a pressure history on each valve, allowing adequate modeling of steam flow through the valves.

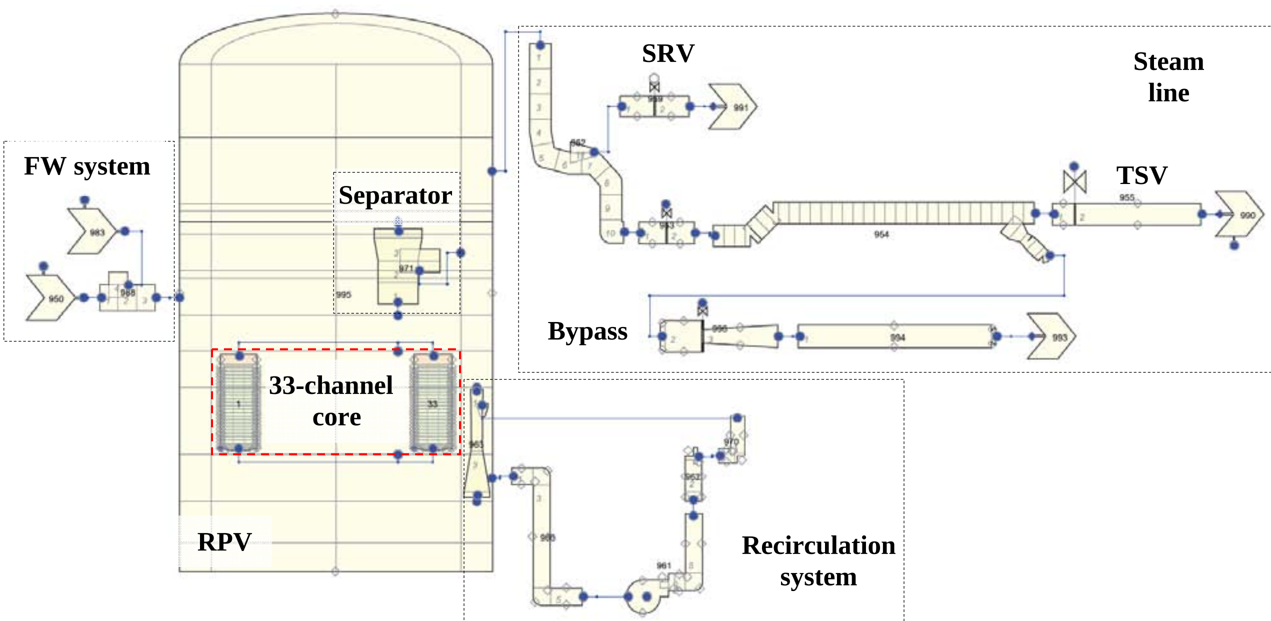

Figure 1. Schematic TRACE model for PB2 TT2

The schematic TRACE thermal-hydraulic model for PB2 TT2 is shown in Figure 1. This model is similar to the TRAC-M model reported in Ref. [6]. However, only two radial rings are used in the RPV nodalization instead of the four rings employed in the TRAC-M model.

The neutronic core model is based on the S3K standalone 764-channels model described in details in Ref. [5]. The S3K core model relied on Studsvik's codes to generate the cross section data, and the history data. All data pertaining to the core (loading, assembly dimensions, pin enrichments, etc.) were taken from the specifications. The cross section data for all lattices present in Peach Bottom cycles 1 and 2 were generated by means the lattice code CASMO-4 ("L"-library). A SIMULATE-3 model for Peach Bottom was set up and depleted for two cycles until the end of cycle 2 corresponding to the point at which the turbine trip transients were conducted.

4. RESULTS

The following boundary conditions were applied in the PB2 TT2 calculations: (a) valve positions versus time for the turbine stop valve and turbine bypass valve, (b) measured feedwater flow versus time, (c) constant feedwater temperature, (d) scram activation time. The bypass valve flow area was adjusted to match the bypass steam flow-rate with the one predicted by the Exelon simulation [13]. In addition, the jet pump parameters (flow areas and loss coefficients) were adjusted to get the correct drive flow and *M* and *N* ratios, as given by the test specification. The simulation was done with the "flat" coupling option.

The initial steady-state conditions in comparison with the S3K standalone results and the measured data are given in Table 1. The TRACE/S3K steady-state parameters are close to the measured values. The average core void fraction is the same as the Exelon value. The initial average void distribution shown in Figure 2 is also close to the Exelon data.

Table 1 Steady-state conditions

Parameter	S3K [5]	TRACE/S3K	PB2 TT2
Power, MWth	2 030	2 030	2 030
Core flow, kg/s	10 445	10 445	10 445
Bypass flow, kg/s	781.4	896.9	841.68
Bypass flow fraction, %	7.48	8.59	8.06
Core inlet temperature, °C	274.67	274.77	274.6
Core inlet subcooling, kJ/kg	48.3	48.68	48.005
Dome pressure, MPa	6.7989	6.7988	6.7985
Core inlet pressure*, MPa	6.9684	6.9266	-
Core outlet pressure*, MPa	6.8640	6.8341	-
Core pressure drop, MPa	0.10443	0.0925	0.08357
Core average void, %	30.4	32.0**	30.4
k_{eff}	0.99150	0.99521	-
Feedwater flow, kg/s	980.8	979.2	980.3
Jet pump drive flow, kg/s	2 964.7	2954.3	2 871.5
Jet pump M factor	2.523	2.536	2.638
Jet pump N factor	0.17	0.17	0.17

*Calculated at the node edge.
**Exelon average void is 32.0%.

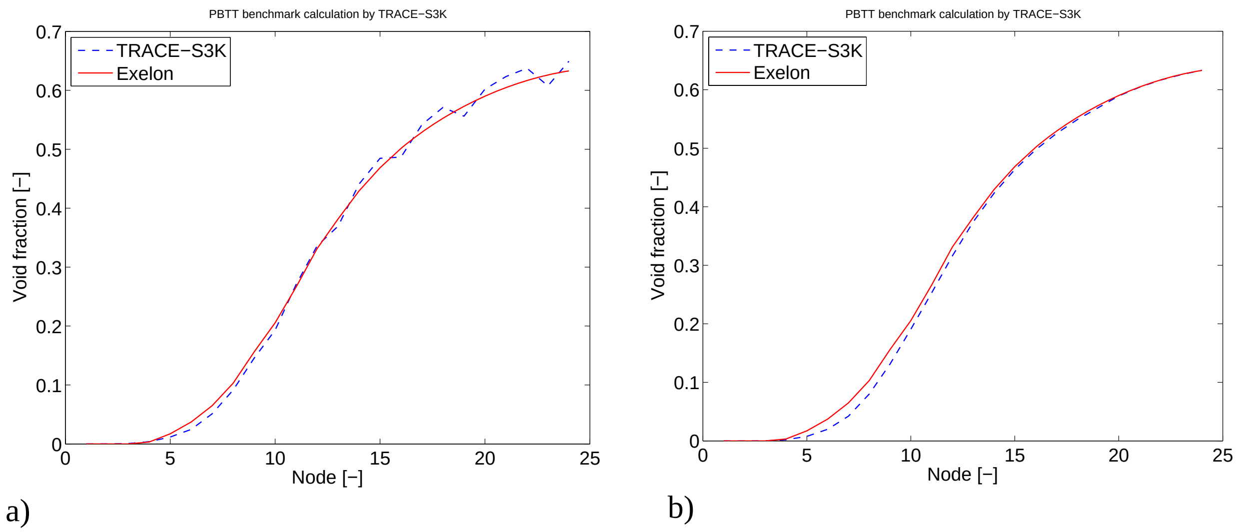

Figure 2. Initial core void profile: a) K-loss factors according to the specifications, b) K-loss distributed uniformly along the channels

The sharp depressions in the void fraction obtained by TRACE/S3K (Figure 2a) occur at the spacer-grids locations. The effect of the spacer-grids is modeled by implementing local friction factors at the corresponding axial locations. The reason for the behaviour of the TRACE/S3K void fraction is related to the modeling of the slip ratio (ratio between steam and liquid velocities) within TRACE. As a matter of fact, in proximity of the spacer-grids, TRACE predicts a sharp increase of the slip ratio, which results in a decrease of the local void fraction. If the frictions factors are distributed uniformly along the channels, the void fraction would monotonically and smoothly increase (see Figure 2b). However, it would result in a decrease of the total pressure drop across the core. The initial distribution of average axial power profile is plotted in Figure 3.

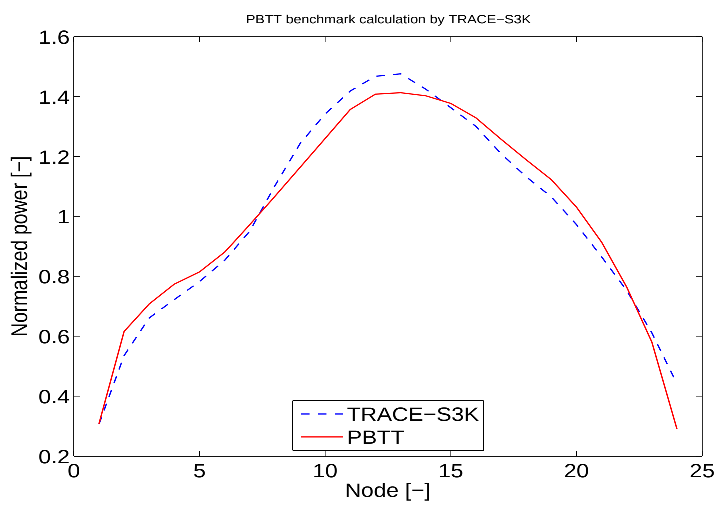

Figure 3. Initial core axial power profile

The transient is initiated by closing the turbine stop valve (closing time interval is 0-0.096 s). The pressure wave travels from the turbine stop valve to the core, resulting in an increase of the dome pressure as illustrated in Figure 4 (pressure peak around 1s). The consequent collapse of voids produces the power excursion shown in Figure 5. The power increase is stopped by negative feedback at 0.77 s and then by scram, which is activated at 0.78 s. The scram signal was delayed from 0.75 s to 0.78 s in order to ensure that there was no impact of the scram on the power peak. During the first 1.0 s of the transient the maximal time step was set to 0.00025 s both for neutronics and thermohydraulic integration. The rest of the transient was calculated with the maximal time step equal to 0.025 s.

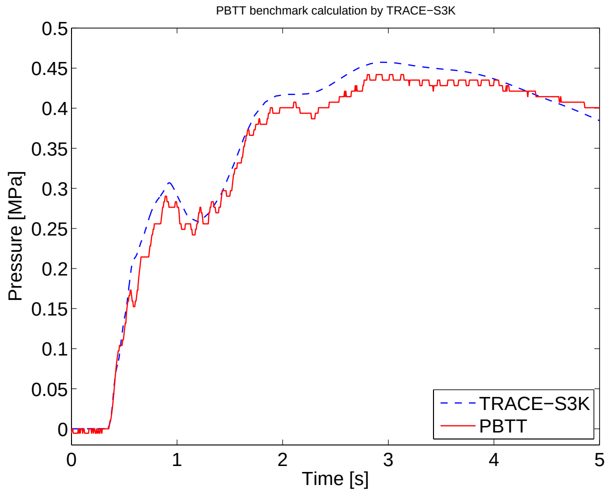

Figure 4. Dome pressure change

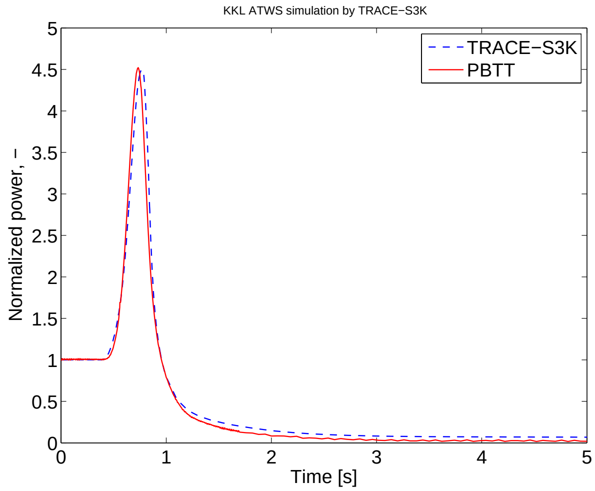

Figure 5. Relative power

Simultaneously with the closure of the turbine stop valve, the turbine bypass valve begins to open (opening time interval is 0.06-0.732 s). The resulting steam flow through the bypass valve is shown in Figure 6. The bypass valve opening recovers the steam line flow-rate at a level of approximately 500 kg/s. After 3 s of transient, the steam line exit flow-rate (sum of the flow-rates through the turbine and turbine bypass valve respectively) is almost equal to the steam line inlet flow-rate (the steam flow from RPV) as drawn in Figure 7. More details of the system dynamic response are given in Ref. [14].

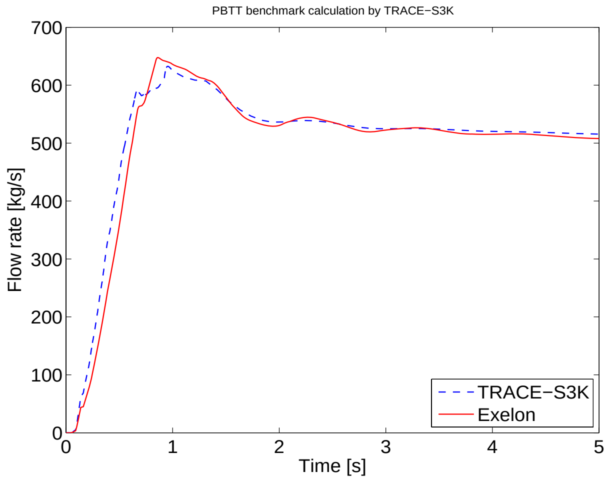

Figure 6. Turbine bypass valve flow rate

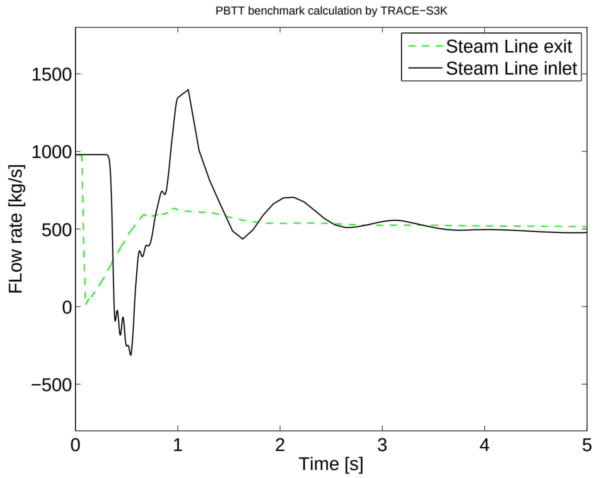

Figure 7. Steam line flow rate

The typical CPU time (Intel® Core™2 6600 processor) was 7.5 minutes for initialization and 70 minutes for 10 seconds of the transient. Note that the restart option is not yet implemented for the TRACE-S3K code.

5. SENSITIVITY CALCULATIONS

As noted in Section 2.3, there are three coupling options available in the TRACE/S3K code: nodal, flat and plenum options. The results above were obtained using the flat options. The nodal option may produce better results when the number of channels is limited since the current nodal powers are utilized to estimate the true three-dimensional density and fuel temperature distributions in neutronics simulation.

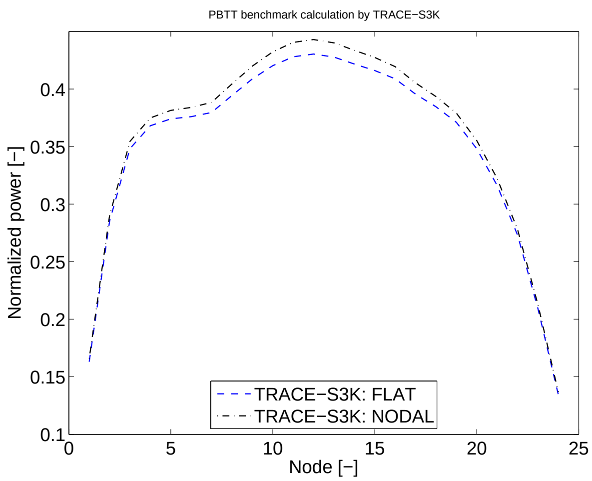

Figure 8. Initial axial power profile for selected fuel assembly

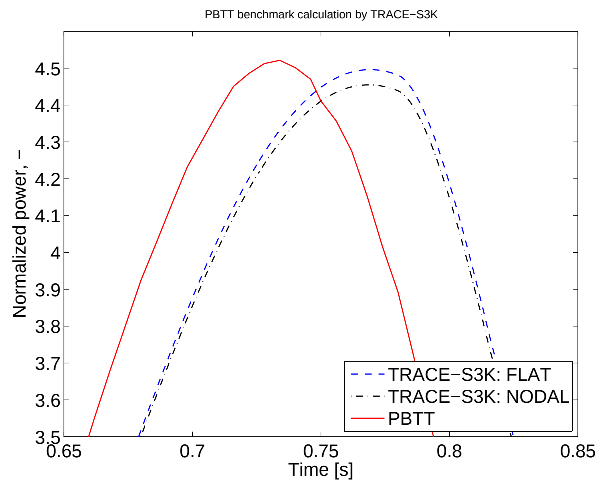

Figure 9. Relative power for different coupling options

The base case results obtained with the flat coupling options were compared with the results calculated with the nodal option. The differences are rather small. The initial axial power profiles for the fuel assembly with maximal radial power obtained with the flat and nodal options are nearly the same. The largest differences are observed in correspondence of the fuel assembly with minimal radial power; these differences are however rather small, as shown in Figure 8. The maximal power during the transient is slightly higher for the flat option as illustrated in Figure 9. However, these differences do not affect much the transient results.

The reference reactor core includes 764 fuel assemblies collapsed into 33 channels. Up to 76 fuel assemblies are collapsed in a single channel. Such a coarse mapping may affect the transient simulation results as was discussed in Ref. [15]. In order to investigate this effect, the basic model was refined into 106 channels. Each channel of the original 33-channels map was splitted into few channels thus the single channel of the developed 106-channels map includes 4-8 fuel assembles with similar power. The comparison for steady-state conditions is given in Table 2.

Table 2 Steady-state conditions for 33- and 106-channels models

Parameter	33 channels	106 channels	Difference
k_{eff}	0.99521	0.99428	93 pcm
Axial offset	5.80%	5.18%	0.62%
Peak source	2.22377	2.16017	2.9%

The insignificant changes are observed with regards to the average core axial power profiles when employing the 33- or the 106-channels model, as shown in Figure 10. The difference in the maximal relative power is also small (Figure 11). The impact to the pressure distribution/history is minimal. The increase in total CPU time was about 42% for the 106-channel model in comparison to the 33-channel model.

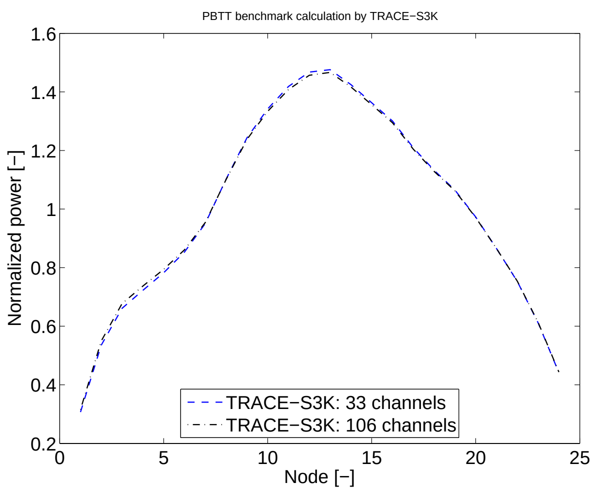

Figure 10. Initial core axial power profile for 33- and 106-channels models

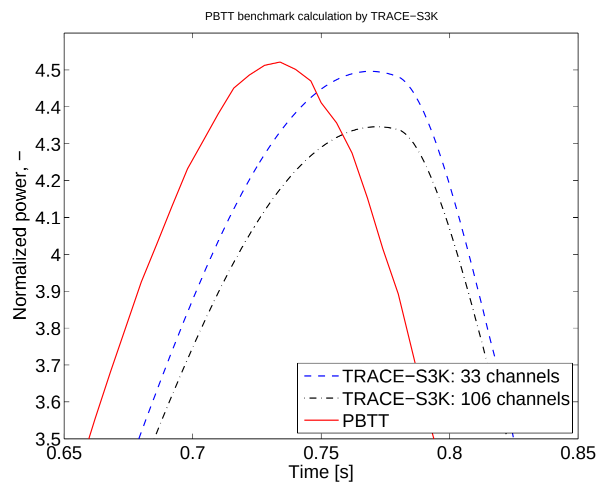

Figure 11. Relative power for 33- and 106-channels models

6. CONCLUSIONS

The newly developed coupled code TRACE/S3K was tested by simulating the NEA/OECD Turbine Trip benchmark. The comparison with the experimental data shows that the TRACE/S3K code reproduces well the main transient phenomena, namely, the pressure wave propagation, void collapsing and core power response. No noticeable differences were found using the "flat" and "nodal" coupling options. The implementation of "plenum" option and restart option is under way. The further code verification will include the simulation of the PWR MSLB benchmark and comparison with the Swiss reactors plant data.

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